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Options for an Early LFR Technology Demonstrator

"High-Confidence Design Options for Lead-Cooled Fast Reactors"

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Summary

The Lead-Cooled Fast Reactor (LFR) is one of the concepts being explored within the DOE-NE Generation IV nuclear reactor R&D program. Because the fast reactor R&D was placed on a long-term (30 year) schedule for research, development and demonstration, the LFR team aimed at a set of technology advancements to provide an innovative and high efficiency advanced power plant. This focus has led to the misconception that any LFR must be decades from readiness. In reality, it is possible to build a LFR demonstration reactor in the near-term with suitable selection of performance parameters and objectives.

This study evaluated the possibility of proceeding with near-term development of an LFR demonstration reactor based on the above observations, and identifies the characteristics of two possible options that depend on the objectives for the demonstration reactor and the funding available. “Near-term” means that the facility could be placed in operation in the 2016 to 2022 time period depending on the government’s development plans. The 2016 date might be achieved if the LFR was selected to play a significant role in the GNEP and there was an urgency identified for the project to be initiated before 2008, thus allocating eight years for R&D and construction. The 2022 date recognizes that there is currently no urgency identified for such a facility and allows six years for continued research and development of the LFR options.

Two options for a development program have been considered. The option with the least programmatic risk is a reactor that is similar to the SBVR-75/100, using oxide fuel, except the power level is reduced to 60 MWt and natural circulation of the LBE is employed with a saturated steam cycle. It would be designed to support development of more advanced materials and fuel but the level of experimentation would be limited by the core size and access available. A higher power level option using once-through superheated steam cycle was considered. This option provides a larger core with potentially more experimental capabilities and uses a super-heated steam cycle that is more efficient than the saturated cycle. Both options use currently available materials and would be capable of operating at least 15 years. There are many specific design issues to decide but they would not be expected to impact the ability to construct either plant with a high level of confidence about its successful operation. There is also the possibility of using SCO_2 as the power conversion system if it is developed in parallel with the reactor for use at low temperature. It is estimated roughly that the smaller plant could be designed, constructed and operated for 15 years for less than $700 million.

1.0 Introduction

The Lead-Cooled Fast Reactor (LFR) is one of the concepts being explored within the DOE-NE Generation IV nuclear reactor R&D program. Because the fast reactor R&D was placed on a long-term (30 year) schedule for research, development and demonstration, the LFR team aimed at a set of technology advancements to provide an innovative and high efficiency advanced power plant. For example, the SSTAR design developed in FY 2005 is based on a reactor outlet temperature that requires fuels materials development and an innovative super-critical CO_2 (SCO_2) Brayton cycle power conversion system that also requires R&D. This has led to the misconception that any LFR must be decades from readiness. In reality, it is possible to build a LFR demonstration reactor in the near term with suitable selection of performance parameters and objectives. This report evaluates the technical issues that would accompany such an early LFR demonstration, and considers the potential benefits from an early LFR.

In his State of the Union Address the President announced the Advanced Energy Initiative (AEI), and in his February 18, 2006 radio address he outlined the Global Nuclear Energy Partnership (GNEP), the nuclear research and development component of the AEI. He also identified in his radio address that, "As these technologies are developed, we will work with our partners to help developing countries meet their growing energy needs by providing them with small-scale reactors that will be secure and cost-effective." He also identified that we would work with countries that have advanced civilian nuclear energy programs, such as France, Japan and Russia.

Many countries including Japan, Russia, South Korea, Argentina and South Africa are developing small reactors for use in various applications, including expansion of nuclear power in developing countries. Small reactors are most suitable for use in remote locations and for the production of fresh water. In general, water distribution over great distance is more difficult and expensive than electricity distribution and the size of local region water requirements match well with the small reactors (10-100MWe) power levels. Russia has developed a small Lead Bismuth Eutectic (LBE) cooled reactor design, the SVBR-75/100, and Japan has developed the sodium cooled Super Safe Small and Simple (4S) reactor. These are intended for export to developing countries as well as for use in remote locations. Based on this experience and the international need for small-scale reactors, the GEN-IV program includes a Lead Fast Reactor (LFR) project. Specifically, the LFR project is developing the Small, Secure, Transportable Autonomous Reactor (SSTAR) for export to developing countries that need economical, clean, safe energy, while reducing the risk of nuclear proliferation.

The LFR project activities are based on the premise that these small-scale reactors will not likely be needed or developed until 2030 or later. This study evaluates alternatives to the current plans for development and testing of the SSTAR. The objective of the study is to establish an approach and basis for early construction of a heavy metal cooled fast reactor test facility. Early construction means that the facility could be placed in operation in the 2016 to 2022 time period depending on the government's development plans. The 2016 date might be achieved if the SSTAR was selected to play a significant role in the GNEP and there was an urgency identified for the project to be initiated before 2009, thus allocating eight years for R&D and construction. The 2022 date recognizes that there is currently no urgency identified for such a facility and allows six years for continued research and development of the LFR options.

There has been extensive R&D and construction experience with sodium cooled fast reactors but relatively little experience with LFRs. Because of this difference in experience it has been assumed that construction of a liquid metal cooled fast reactor in the 2016 to 2022 would require that it be cooled by sodium. This study questions that assumption and identifies approaches for reducing the programmatic risks associated with research, development and construction of a SSTAR technology demonstration facility to acceptable levels on an early schedule.

2.0 Background

LFR systems have been under study by the U.S., Europe, Russia and to some extent Japan and South Korea for more than ten years. The interest in such systems has expanded significantly based on Russian reports about their experience with development of LBE cooled submarine reactor systems. Their subsequent design work on the SVBR-75/100 reactor [1], and the European interest in LBE cooled Accelerator Drive Systems (ADS) [2], has added to the interest. The U.S. has been investigating small LFRs and related technology within the GEN IV program and fuel for these reactors within the AFCI program. The GEN IV and AFCI program activities are planned on the basis of potential construction the first LFR demonstration plant after 2030. Several LFR designs, including the Encapsulated Nuclear Heat Source (ENHS) [3], STAR-LM [4] and STAR-H_2 [5] have been investigated under the Nuclear Energy Research Initiative (NERI) and GEN IV LFR programs, but a preferred design has not been selected.

The most recent pre-conceptual design from the LFR program is the 20MWe SSTAR design completed in 2005 [6]. Table 1-1 provides some of the key design and performance parameters and Figure 1-1 is a schematic of the design. This design was developed on the basis of deployment occurring after 2030, as were the other pre-conceptual designs produced by NERI and GEN-IV. The design parameters have therefore been selected on the basis of having time to develop new fuel and materials that would support innovative design concepts. In the case of the most recent SSTAR design the innovations include using supercritical SCO_2 power conversion and a liquid metal temperature higher than the Russian experience. These advanced features are intended to lower the cost of a commercial facility of similar design. This study identifies alternative design characteristics and development plans that would support a SSTAR technology demonstration project to be completed much earlier than these advanced designs. The study focuses on the SSTAR option because it is a small plant and demonstration of the reliability and costs of such a plant could be completed on a shorter schedule and at less total cost than alternative larger scale plants. It is also possible that large multi-unit power complexes using small LFRs, such as SSTAR, or intermediate LFRs, such as STAR-LM, may prove to be competitive with large (≈1000MWe) size plants [7].

Table 1-1 Selected Core Design and Performance Parameters for SSTAR FY 2005 Concept

Design	
Design Lifetime (EFPY)	30
Fuel rod diameter (cm) 1.	2.500
Pitch to diameter ratio	1.185
Fuel smeared density (%)	90
Fuel volume fraction	0.45012
Cladding volume fraction	0.14570
Bond volume fraction	0.05001
Coolant volume fraction	0.35416
Performance	
Discharged U (kg)	4462
Net TRU consumption rate(kg/yr)	0.7
Average power density (kW/L)	42
Specific power (kW/kgHM)	7.4
Peak power density (kW/L)	76
Average discharge burnup (MWd/kg)	81
Peak discharge burnup (MWd/kg)	131
Peak fast fluence (10^{23} n/cm^2)	4.01
Maximum reactivity swing (%□□)	0.32
TRU feed (kg)	1020
TRU enrichment, w/o	1.7/3.5/17.2/19.0/20.7
Depleted uranium feed (kg)	4953
Total feed (kg)	5972
Discharged Pu (kg)	865
Discharged minor actinides (kg)	134

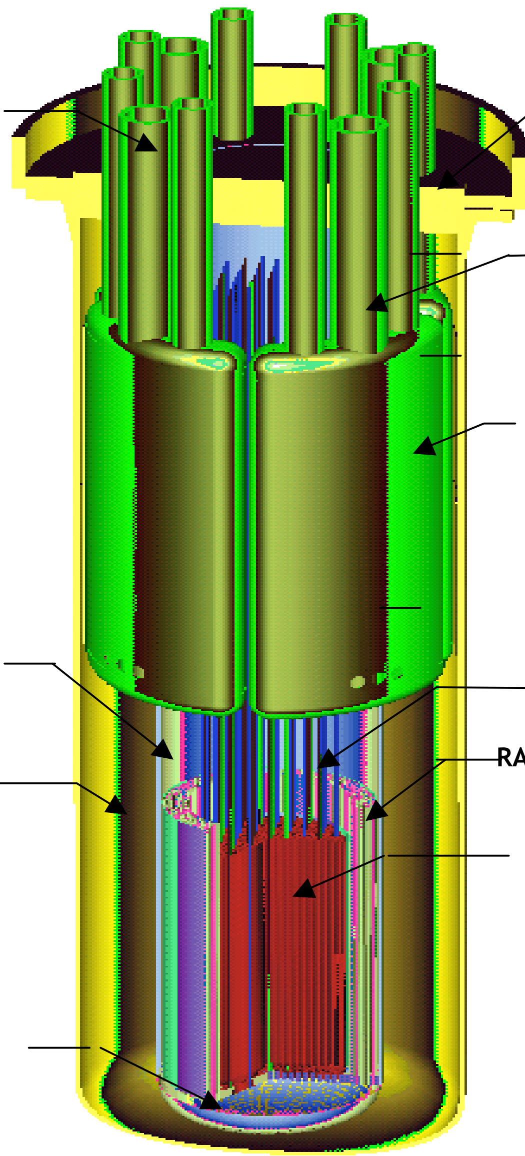

Figure 1-1
SSTAR FY 2005, 20MWe Reactor for use with a SCO_2 Power Conversion Concept

2.1 Objectives

The objectives of this study are to identify the facility design parameters for an SSTAR technology demonstrator that could initiate construction at the earliest time possible, with an acceptable level of technical and programmatic risk. The R&D elements and schedule necessary to control the risk are also identified. The key to accomplishing this is selecting design parameters and characteristics, and a program of testing that are not excessively demanding but provided valuable demonstration of the SSTAR technology

A set of general objectives and requirements that must be met are identified. These constrain the range of options to be evaluated and also help identify the research and development that must be completed to control the risks associated with design, construction and operation of the facility. Thirteen design, operational and programmatic parameters have been identified and a range of options in each topical area is discussed and evaluated. One of the more important of these parameters is the scope of technology demonstration that the facility is required to complete. Such a first-of-a kind facility is usually a compromise between demonstration of developed technology, provision for testing to develop emerging technology, and technical risk. The evaluations in this report are used to identify facility configurations and an R&D program supporting the early construction with acceptable risk.

2.2 Facility Scope

There is a wide range of commercial configurations that have been envisioned for the small LFRs, different power level modules, multi-module plants and high temperature high efficiency modules like the current FY 2005 SSTAR pre-conceptual design. All of these concepts would benefit from an early LFR technology demonstrator that may not be prototypical of any of these eventual commercial configurations but addresses technology important to them all.

Facility and test program scope can vary considerably. The objective is to identify a program for a reactor facility that can be conducted in about the same time period as that for a sodium cooled reactor. This means that it will rely as much as possible on the sodium cooled reactor experience. And given this situation it will mean that the selected program may not address all of the technology that could be important to an optimized prototypical design. For example, the current SSTAR design requires development of the SCO_2 power conversion system and materials that permit operation at peak clad temperature of 650ºC. Although this may be possible for a facility is to be constructed in 2030 or later, permitting the development of new materials and systems, it is not possible to do on the proposed early schedule. The high temperature and SCO_2 energy conversion system are beyond current LFR and sodium reactor experience and imply development programs that will require considerable time to complete. This is especially true for the high temperature. It may be possible to develop a lower temperature SCO_2 system in parallel with a near-term reactor design, however this would introduce additional programmatic risk.

In addition to the technical issues, there are cost and program schedule issues. The duration and scope of the test program will impact both the cost and schedule. A facility that includes power conversion equipment will cost more than one that dumps heat to the air. On the other hand, inclusion of the power generation components will mean that there will be income from the electricity generated and demonstration of the interfacing of the reactor system with a power generation system. The cost of a 100MWt facility also will be more than a 50MWt one, therefore selection of the power level is an important consideration, and operating the facility for 20 years will cost more than conducting a 10 year program. It is desirable to select the programmatic parameters so that the technology demonstration will provide a substantial data base for design, construction and operation of a competitive commercial plant while meeting the early schedule and minimizing the cost. Section 3.0 discusses each of primary parameters associated with selection of LFR technology demonstration and Section 4.0 identifies several program options that are available based on the evaluations in Section 3.0 and outlines facility and program options that could be implemented on an early schedule.

2.3 "Must Have" Requirements

There are several requirements and assumptions used in the study that must be satisfied by the reactor design and test program. Some of these are obvious while others deserve further evaluation under a more comprehensive planning effort

1. The facility is expected to be constructed on a government site, with the nuclear energy research laboratory at INL being a logical choice. (This assumption supports assumptions 8 and 9 below. An alternate site might result from the formal site selection process.)
2. The reactor must be cooled by lead or LBE.
3. Fuel will be designed for 160,000 hours of effective full power operation, but at most will experience only 80,000 hours during the test program. (The longer life is consistent with the SSTAR objectives. It is assumed that demonstration of the shorter fuel life would be an accomplishment sufficient to justify the investment.)
4. The steady-state peak cladding temperature must be less than 550 ºC which corresponds to an average core outlet temperature between 435 and 480 ºC.
5. The materials of construction, including fuel and cladding must be standard materials approved for the conditions of use or the program must complete a demonstration of their adequacy..
6. The facility shall include instrumentation and servicing equipment needed to validate the technology but not planned for use in the commercial plant, such as fuel handling and storage equipment, additional neutron, temperature and flow instrumentation.
7. The reactor core shall consist of hexagonal assemblies of fuel pins on a triangular pitch, and will include locations that support flow and transient testing of several LFR fuel design concepts.
8. The reactor shall be designed and operated so that following 30 days of shutdown the spent fuel assemblies can be removed and stored indefinitely in dry casks. The casks will be suitable for fuel transport on INL roads closed to the public. (This assumption may not be prototypical of the commercial design but is important to implementation of the early test program.)
9. It is assumed that existing fuel examination facilities at INL are adequate to support the PIE of fuel and materials irradiated in the facility.
10. The program will include comprehensive scope of transient testing, including for example, loss of heat sink without scram.
11. The design will provide in-core positions for irradiation testing of advanced structural and fuel materials.
12. The program will include in-service inspection and maintenance comparable to that anticipated for the commercial installations.

3.0 SSTAR Parameter Options

This section evaluates the viable range of parameters that could be applied to the early technology demonstration facility.

3.1 Scope of Demonstration Project

The purpose of a reactor technology demonstration project is to demonstrate the viability of technology for use in a future commercial plant design, construction and operation. Ideally, one would like to build and operate the prototype of the plant that is thought to be commercially viable. This is usually not possible because of the size and cost of the commercial plant and perhaps the developmental status of the materials used in the systems, structures and components. Historically, a reactor demonstration project has not been prototypical of the commercial design and has been smaller to reduce the cost. In the case of sodium and water cooled reactors, many small scale plants that demonstrated much of the technology, particularly the fuel, materials, operational control and safety, typical of the commercial design have been completed. Large-scale plants, more prototypical of the commercial designs, were then constructed. The commercial size plants have also evolved as experience is gained. This has been the development path followed by both the water and sodium cooled reactor development.

In the case of the small LFR, an option is to build the first prototypical reactor so that it is very close to the projected commercial design. This is a unique approach that is possible for small compact designs. It is the also the approach proposed by GE for its sodium cooled reactor PRISM [8]. The reactor module used in the multi-module commercial PRISM plant design was proposed as the only reactor demonstration needed to support multi-module commercial design. Like PRISM, the small LFR could be used in a multi-module plant but its target market has been the developing countries and remote locations that can use power plants in the range of 10-100MWe provided by a single module. Because of the wide range of potential commercial application of a small LFR like SSTAR, establishing the scope for a technology demonstration requires considerable engineering judgment. This study is based on selecting design parameters and features for the demonstration facility that can be constructed and operated in the near-term and that will provide significant technology support to design of future commercial small LFR plants. This means that the demonstration may not be prototypical of the commercially viable design but will contribute significantly to reducing the uncertainty about constructing and licensing these designs.

The most competitive commercial design of a small LFR is yet to be determined. The Russians SVBR-75/100 design is estimated to be competitive with their VVER-1000 and operates on a saturated steam cycle with a peak fuel cladding temperature of 600 ºC. They believe that it may be possible to improve this design to operate at a 650ºC with superheated steam and become even more economical. The SSTAR design outlined in Section 2 is also estimated to have a peak cladding temperature of 650 ºC and may be an even more economical design because of the high efficiency of the SCO_2 power conversion. There is currently no data to support long-term steady state operations at this temperature and substantial fuel and materials research is needed to support design and licensing of an LFR with peak cladding temperature of 650 ºC. Thus, a lower temperature design is a must for an early technology demonstration. Lower temperature designs, preferably below a peak cladding temperature 550 ºC, may also be economically competitive, especially in remote locations and developing countries where the cost of energy is high.

Reactor

The technology demonstration must obviously include a reactor with a core very similar to that planned for commercial use. The small LFRs considered to date have been the pool type. There is no compelling reason to consider a loop type plant given the experience and studies completed to date. The reactor vessel includes all the primary coolant and the heat exchanger for the power conversion fluid. It may also include the primary coolant circulation pumps if these are used. Equipment used to control, monitor and service the reactor must be included. This equipment may be prototypical of the commercial designs but deviations should be permitted to reduce cost. The reactor facility must include an acceptable means for controlling the accidental release of hazardous material.

Balance of Plant

The balance of plant equipment shall be sufficient to support a full scope of operational and transient testing. The most cost effective way to achieve this may be to include the power conversion equipment appropriate for a commercial design. This equipment and the design interface with heat exchanger in the reactor shall be based on proven methods but may require custom design.

Auxiliary Systems

Systems necessary to refuel, maintain and inspect the facility must be provided but these may deviate from the commercial design to reduce cost or support early implementation. Specifically, the core design and fuel handling shall be accomplished using individual fuel assemblies and proven methods for conducting fuel or test article insertions and removals. Core radial restraint and inlet plenum design shall be consistent with this configuration.

Materials and Fuel Testing

Since the constraints applied, such as low temperature, and the use of developed materials are not necessarily optimum for commercial designs the facility, including the reactor core and core access equipment should provide the capability to insert and remove samples into the select core positions. Materials placed in these positions should only be cooled by the primary coolant. The reactor is not expected to accommodate independently cooled closed loops.

Operations

The operations and security staff shall be consistent with DOE requirements for a test reactor. The control room and instrumentation and controls shall support the test operations and are expected to deviate from the commercial design.

Decommissioning
The coolant and fuel used during the demonstration program shall be removed and stored or disposed of at appropriate DOE or commercial sites. Structures will be decontaminated or removed so that unrestricted access to the area is permitted.

In addition to the above items there is the question of how long the facility must be operated to gather the information needed to support the viability of the commercial design. The plant life for the commercial designs is targeted for 30 years, or more with fuel cartridge reloading. For a technology demonstration plant to be useful to the overall program it would need to be capable of completing its technology demonstration in considerably less time, say 15 years. It would be desirable to have significant results in even less time. This means that the methods for accelerated testing of some aspects, such as integrated flux and fuel burnup will need to be accelerated and methods for extrapolation of partial life testing to 30 years must be identified. For example, test fuel assemblies with higher than prototypical enrichment may be used or the reactor may be designed for a higher power density than planned for the commercial reactor. Such design features must be considered in the light of the desire to do a full scope of safety testing. It may require a program that includes both a prototypical power density core and a smaller high power density core to obtain the desire technology demonstration. Design of such a test program is beyond the scope of this study but it is judged that design such a program should be possible without impacting the ability to implement and early demonstration facility. That is, the information important to establishing the viability of the commercial designs can be obtained within the constraints needed to permit a low risk early design.

3.2 Power level

Selecting the power level does not generally impact the possibility of performing an early technology demonstration but it will impact the cost and schedule. Ideally the demonstration facility should be built and operated at the power level identified for the plant envisioned for commercial deployment The power level for commercial designs has been considered to be between 10MWe and 100MWe. Thus there is a wide range of power levels that could be acceptable to the technology demonstration. There is a power level below which the technology demonstration objectives would not be achieved. For example, the core should be of sufficient size to exhibit all the reactivity coefficients present in the commercial design. Such features as core support and core restraint should be of a type that is thought most appropriate in the commercial design. The 10MWe size plant is probably the minimum size that will meet the objectives but it is likely that a larger size plant might be desired in order to better serve the full range of commercial plants. A larger size plant will also facilitate incorporation of testing locations in the core for fuel and materials that are preferred in the commercial designs. Based on the core layout for the design developed in 2005 it is judged that a 60MWt core might be a reasonable size for meeting these objectives. Increasing the power level above the minimum will increase the cost but it is not expected to increase the programmatic risk or impact the possibility of construction and early technology demonstration. These are influenced by other parameters discussed and evaluated in this study.

3.3 Temperature

The temperature is the single most important parameter impacting the ability to implement an early LFR technology demonstration reactor facility. Other design parameters such as core power density, linear power and fuel burnup are less than values that have been fully demonstrated as acceptable in sodium cooled fast reactors.

One of the advantages of LFRs is the potential to run the reactor at temperatures that are higher than acceptable for sodium cooled reactors. Core outlet temperature of 800 ºC has been proposed for some designs [5]. However, the materials to implement such designs are not readily available or difficult to economically implement in commercial designs. These materials are typically high nickel alloys and are not suitable for use in lead or LBE due to the high solubility of nickel in lead and bismuth. Higher temperatures are a challenge to both the structural and fuel materials that are available to support an early demonstration. Higher temperature also exacerbates the corrosion potential of lead and LBE and makes it more difficult to use demonstrated means for controlling corrosion of iron based structural materials.

The early Russian designs appear to have kept the peak cladding temperature below 550 ºC, which is judged to be a temperature at which LBE corrosion is clearly controllable, at least for operating periods up to ten years and maybe longer. Recently the Russians have advertised designs with peak cladding temperature at 600 ºC for up to 30 years and think that they may be able to support 650 ºC in the future. The basis for these improved performances has not been explained. There currently is no basis for this high of a temperature for an extended period. Thus, to support a low risk project the peak cladding temperature should be kept between 550 ºC.

3.4 Coolant

Because of the limit on the outlet temperature, the only viable coolant option are LBE (55%Bi 45% Pb), used by the Russians or another lead eutectic with a melting temperature less than about 300 ºC, such as PbSn eutectic (melting temperature, 183.3C). It is desirable to have a good margin on t-melt and also have a reasonably large delta-T across the core.

The principle advantage of LBE and other eutectic alloys over lead is their lower melting temperature. The fact that the Russians continue to advocate LBE following more than 80 reactor years of experience provides a testimony for this coolant in spite of the fact that it produces Po-210 and is potentially more corrosive than lead. The extensive Russian and European experience with LBE and the generally good U.S. experience with operating the DELTA loop at LANL, represents a substantial data base for design of an early demonstration plant. The Po-210 issue is an operational concern but may be present with all lead alloys since Pb-210 is converted to Bi-210 and then Po-210 with several neutron captures, thus there will be Po-210 at some level in all lead alloy coolants. It will be much larger with LBE, but the Russians continue to claim that worker safety can be maintained without too onerous of controls. Recent studies at MIT of high temperature lead cooled reactors indicates that maintenance and repair operations may not have significant advantages over LBE cooled reactors as relates to Po contamination.

If a coolant other than LBE were selected it would mean that there would be additional effort required to obtain the necessary data base and experience. This additional effort would not be expected to be on the critical path since much of the testing would be conducted out-of-pile. It would add non-trivial costs, in the order of several tens of millions of dollars. Because of the higher melting temperature of lead and other lead alloys, and the 550 ºC (435-450 ºC average core outlet) limit on peak cladding temperature, the core ΔT would be low and would lead to a high flow requirement.

3.5 Fuel

3.5.1 General

The fuel is a system that includes fuel, cladding and a bonding material to improve heat transfer. In this section the fuel and its compatibility with the bonding material is evaluated. The Russians have used UN fuel in their designs and this is the preferred fuel material for the commercial design. The high density and conductivity of UN support very attractive core designs from the standpoint of both performance and safety. The U.S. has also developed UN fuel for use in space power reactor systems. However UO_2 and the ANL developed metal fuel are alternatives that may be more readily available. These alternatives are evaluated in this section and a recommendation made.

3.5.2 UN Fuel

UN fuel, although the preference for the commercial system designs, it will require development testing that is likely to extend the cost and schedule parameters beyond those associated with an early demonstration project unless extensive qualified data were to become available from Russia. There may be extensive and useful data from Russia that would benefit an early demonstration but it is questionable whether the scope of testing, and quantity and quantity of the data records, would adequately support the U.S safety review process. A conservative fuel design approach for the driver fuel that: (1) excludes use of Pu or MAs to improve manufacturability, (2) uses low smeared density to eliminate concerns about pellet-clad interactions, (3) includes an excessively large fission gas plenum volume to accommodate gas release, and (4) uses thick cladding to address concerns about corrosion may be an approach to address the technical risk. Even with this conservative design approach, there will be a need for irradiation data to demonstrate the safety margins under the design basis accident conditions and possibly the severe accident conditions in order to obtain safety approval. There is also the problem with establishing an approved fuel manufacturer.

Naval Reactors recently faced the issue of selecting fuel for use in the Jupiter Ice Moons Orbiter (JIMO) mission they were planning with a gas cooled fast reactor operating 20 years. Following considerable evaluation they selected UO_2 over UN because the development time required and risk associated with developing and qualifying a supplier of UN fuel. This fuel was going to be operated at higher temperature and for a longer period of time than will be required in a LFR technology demonstration facility, but it does provide some warning about potential difficulty with developing UN. They reviewed all the data available in the US, Russia and Japan and concluded it was not sufficient nor was there time to develop the data to support the early launch schedule required by JIMO. Thus, developing a U.S. supplier of UN fuel on an acceptable schedule is also problematic.

3.5.3 Metal Fuel Option

Argonne has developed and used U-Pu-10Zr alloy fuel for many years in the sodium cooled EBR-II reactor. This fuel has also been irradiated in FFTF and selected for use in the sodium cooled Super Safe, Small and Simple (4S) and GE PRISM reactor designs. This fuel and UN fuel are similar in density and thermal conductivity. High fuel conductivity results in a fuel temperature at full reactor power that is only about 50 ºC higher than the coolant temperature. This is most important to achieving a demonstrably safe response of the reactor to postulated unprotected (transients that assume failure to scram) accidents. The gain in the reactivity as the fuel is cooled is small and much less than will occur in core that uses oxide fuel. (UN fuel is predicted to provide a response to these postulated accidents that is similar to that of metal fuel.)

The metal fuel used in sodium cooled reactors is bonded to the cladding with sodium. Sodium is not acceptable for use as a bond in LBE reactors because the sodium and LBE have exothermic reactions and will lead to stripping of the oxide protection from the cladding. An effort would be required to demonstrate the suitability of lead or LBE to bond the metal fuel. Bonding with He is another possibility but the resulting higher fuel temperature may impact both the safety and design performance in an unacceptable manner.

In all reactors there is a concern about operating with fuel that has a cladding breach. When the coolant comes in contact with the fuel it may erode the fuel and disperse it in the coolant volume. Oxide fuel in LWRs is vulnerable to this problem and LWRs typically do not run for long with fuel that has a major breach. Metal fuel exposed to the flowing liquid lead or LBE may present a more severe problem than is the case with LWRs, in that the fuel may dissolve in the lead over time and not be recoverable. The requirement to detect and remove failed fuel may also be most demanding in a metal fueled core.

There is considerably more performance and safety data relative to the metal fuel available in the U.S. than there is for the UN fuel. Based on the NRC comments from the PRISM pre-application review and the termination of the IFR program planned to support licensing of PRISM there is a need to restore the fuel performance analysis tools and complete the planned development program [9]. Much has been done by ANL to maintain and improve many of the tools, but not all, and therefore significant effort may be required in this area if the reactor were to be licensed as a power reactor. Like the UN fuel, developing a supplier of qualified metal fuel may require an extensive program, even though there has been fuel manufactured and tested in FFTF. There is no commercial vendor experience with manufacturing the ANL developed metal fuel. The facilities at INL could possibly be used and production completed there but the schedule for this may be limiting and if licensing is required NRC qualification of this facility may be as difficult as for a commercial vendor. The time required to do this may be lengthy and not consistent with an early demonstration.

3.5.4 Oxide Fuel Option

The major reason for considering oxide fuel is that it is the most mature and experience based fuel form for power reactors and the only power reactor fuel for which there are several qualified industrial suppliers. Thus, it is very likely that an NRC qualified fuel supply can be established on the schedule that would support an early demonstration LFR. This fuel form has been used in most of the fast reactors operated in the world, with FERMI-I and the ANL designed reactors built at INL being the exceptions.

The major penalty accompanying UO_2 fuel is its low conductivity and the resulting high temperature. In spite of this property it has been used in many fast reactors. However, oxide cores do not have the safety response to postulated unprotected accidents that metal and nitride cores provide and they may require more engineered safety features, including the need to establish capability to withstand energetic core disruption. It is very desirable to avoid adding safety features and providing accommodations for energetic core disruption. In fact, the NRC's advanced reactor policy encourages increased passive response to postulated accidents and even though foreign developers of fast reactor prefer oxide fuel the NRC approval of these designs may be very difficult.

Like metal fuel, there is a question about the choice of bonding material for UO_2. Both sodium and He have been used but using lead or LBE may also be acceptable. There is some developmental experience with lead and the Russians have used LBE. If either of these heavy metals were used to bound the fuel it would be necessary to conduct the necessary R&D to support their use. The melting temperature of oxide is sufficiently high and the restructuring that occurs in the irradiated fuel makes improving the gap conductance with a liquid metal bond less important to this fuel than is the case with metal fuel and therefore the HE bonding may be acceptable.

3.5.5 Conclusion

All candidate fuels present problems to proceeding with an early LFR demonstration. UN is probably out of the question for the early demonstration unless there was substantial Russian, and even then qualifying the Russian fuel may require an unacceptably long time. Metal fuel may be acceptable if the INL early demonstration was not licensed and the fuel fabrication facilities could produce the fuel in a timely manner. Oxide fuel, while having available suppliers to support the early demonstration plant presents some safety issues that may add complications and expense.

If significant Russian participation were available, it is possible that the preferred UN fuel could be used and sufficient data would be available to support NRC licensing on a schedule consistent with an early demonstration. There may be a combination of new development work coupled with the existing Russian data base that would also be compatible with an early demonstration schedule. These approaches should be investigated first when establishing an early LFR demonstration project.

There may be an acceptable approach to developing INL as a supplier of metal fuel and addressing the additional R&D without severely impacting the schedule. If the Advanced Burner Reactor uses metal fuel, as currently planned, it would certainly resolve any issues about availability of a supplier. However, there are still issues with the compatibility of this fuel with the lead or LBE and selection of a bonding material. If these are different from the sedum reactor fuel there is likely to be R&D required that would extend the program beyond the earl demonstration time period. Since metal is not the preferred fuel for the LFR trying to use it is unattractive approach since the cost and schedule may not be much different than what could be achieved with UN fuel with minimal Russian data.

Oxide fuel is the only readily available fuel. However, one needs to evaluate the impacts of conducting the early LFR demonstration with a core configuration that does not include the optimum safety characteristics. It may be possible to boot strap the facility from an initial oxide core to the more optimum UN core and therefore complete the necessary R&D needed for the commercial design. It would require some additional effort and possibly safety features, such as a leak tight containment, not necessary in the commercial design that uses UN fuel.

3.6 Cladding

3.6.1 General

Demonstration of the adequacy of a fuel cladding presents the major challenge to an early LFR demonstration reactor. Assuring adequate resistance to coolant corrosion under the reactor temperature and radiation environment is the major uncertainty that must be resolved. The cladding must also be demonstrated to be compatible with the fuel through operational and low probability transients, including those postulated to occur during severe accidents. Using oxide fuel will go along way to resolving the uncertainty about fuel clad interactions.

3.6.1 Cladding Options

It is most important to have cladding that has high resistance to corrosion from the lead or LBE and fission products, low swelling, high-temperature creep strength and is easily fabricated. Table 3-1 lists the possible candidate materials for in fuel cladding. These three materials are all similar in their corrosion resistance if the steady-state peak temperature is kept below 550 ºC. Only the HT-9 has extensive irradiated properties data available, T91is expected to behave similar to HT-9 and EP-823 has considerable Russian irradiation experience. They all are expected to possess desired low swelling behavior. Table 3-2 provides the corrosion parameters and test data available on these materials.

Table 3-1 Fuel Cladding Material Candidates

Material	Cr	C	Si	Mn	W	V	Mo	Ni	Other
HT-9	12	0.25	0.2	0.55	0.5	0.3	1.0	0.55	
T91	8.0-9.5	0.08-0.12	0.20-0.50	0.30-0.60		0.18-0.25	0.85-1.05	≤!0.40	0.06-0.10Nb, 0.03-0.07N
EP823	11.4	0.18	1.05	0.6	0.65	0.4	0.67	0.7	0.2Nb, 0.012P, 0.03Ti, 0.04N

Table 3-2 Corrosion resistance of Material Candidates

Steels	$T_{max,op}$ [^{o}C]	$c_{O,\ op}$ [wt%]	$T_{max,test}$ [^{o}C]	$T_{max,test}$ [h]	Comments
T91	>520, <~600	$5x10^{-7}$-$5x10^{-6}$	520 (600 static)	10,000	For fuel cladding, target window and other high fluence components; may need pre-oxidation for high-T use; oxidation may be severe for long-term use
HT-9	>550, <~600	$5x10^{-7}$-$5x10^{-6}$	550	3,000	Similar to T91
EP823	>550, <~600	$5x10^{-7}$-$5x10^{-6}$	> 550	> 14,000	Russian alloy with developed and deployed for LBE reactors; best performing in side-by-side tests with other F/M steels; may embrittle under radiation at low-T

HT-9 has received more attention in the U.S. because of its role in the sodium cooled reactors programs. T91, is a ferritic steel that may be commercially available and is more similar to the EP823 Russian material used in their LFRs than HT-9. All of these materials have demonstrated good resistance to LBE corrosion if the temperature is kept below 550 ºC. The higher Si content in EP823 gives it the best corrosion resistance and one would expect that T91 is somewhat better than HT-9. The final selection of a material for an early demonstration will depend on the ability to obtain production quantities of qualified material. The ability to do this will be determined by the project commitment to steel producer and this may be on the critical path of the project. It will be necessary to obtain more corrosion and irradiated properties data on the selected material to support the safety review and finalization of the operating limits and fuel life. However, any one of these materials could be used with a fuel replacement plan that would have acceptable risk. In the long term, new materials will be needed for the cladding to improve lifetime and permit higher temperatures. The NEA materials handbook will be issued later this year and will provide support for selecting the material [10].

3.6.2 Evaluation of Options

The ODS materials that are on the market or being developed appear to have good corrosion resistance and would provide improved creep strength at higher temperature. It will require more development to and irradiation in flowing lead to develop the same confidence that exists with the identified candidates. There is also development of bimetal cladding that can be tailored to have material most compatible with fuel on the inside and lead or LBE on the outside. This approach may be necessary for the targeted long life and higher temperature but is not necessary for an early demonstration plant.

3.6.3 Recommendation.

Assuming that there is only minor Russian participation in an early technology demonstration, T91 or HT-9 should be selected as the cladding and a qualification program initiated as soon as possible. The issues associated with obtaining and using EP823 from Russia or developing a supplier in the U.S. should also be investigated along with the data available to support its use. Qualifying a U.S. vendor for this material would probably delay the project to a point that it may not qualify as a near-term system.

3.7 Structure

The material for structural items like the core support and reactor vessels may also be constructed using HT-9 or T91. Depending on the temperature, neutron fluence and mechanical loading 304, 316L and may also be used but these applications but will need to be at temperatures where lead or LBE corrosion is not a problem. These materials may be preferred over T91 or HT-9 because for the lower temperature and low fluence regions where T 91 and HT-9 are more susceptible to embrittling.

3.8 Core life and servicing

It is most desirable to have a core life of 30 years or greater in order to achieve energy cost, proliferation resistance and conveniences that will make these systems attractive for use in developing countries on small grids and remote applications. Features such as whole core, or whole reactor exchange and no on-site fuel storage cannot be economical if the core life is in the range of 5-8 years currently achievable. However, these features would be impossible to demonstrate in the first unit because the fuel data basis is not available and would be difficult to obtain without building a new fast spectrum reactor. The viability of the long life fuel can be obtained with a facility that exchanges fuel more frequently by performing accelerated testing and using selected positions in the core for long life testing. Thus, the demonstration facility can contribute to supporting the viability of the long life core without actually operating in a prototypical mode. In contrast to the commercial design, the demonstration facility must have relatively frequent access to core to permit the needed irradiation testing. Thus the early demonstration facility must include means to get access to the core and exchange assemblies.

Recognizing that the fuel management system of the demonstration facility will not be prototypical of the commercial design permits a number of approaches for fuel management and for fuel test assemblies to be considered. However, it is desirable to have the early demonstration facility to deviate as little as possible from the commercial design. For example, inclusion of features such as rotating plugs in the closure assembly and in-vessel fuel storage locations for spent fuel should be avoided if possible because of the impact they will have on the vessel and control systems design. The fuel management design is an area that will need to have trade studies performed before committing to a design. These should address fuel storage and shipping in addition to the reactor because there may be approaches that provide a better data base for designing the commercial facilities and equipment needed. The demonstration facility is likely to be small enough that the approach used in early GE designs of fast reactors that installed a hot-cell over the top of the reactor may be an attractive alternative. Such an approach might permit the reactor vessel, reactor closure assembly and controls to be prototypical.

The technology demonstration reactor will be the essential test facility for developing qualified long life fuel and should support design of fuel handling and storage whenever possible. For example, an approach to no on-site storage of fuel may be one refueling outage after 30 years in which the fuel assemblies are transferred a short time following shutdown to a few shipping casks that are capable of cooling a few assemblies and are removed from the site immediately. The 4S sodium design includes a concept that uses 6 casks to remove the 18 assemblies, that constitute the core, in a 50MWe plant, from the site shortly after shutdown of the reactor.

3.9 Coolant Circulation

Fast reactors have been constructed with primary coolant loops (loop-type) or with all the primary coolant in the reactor vessel (pool-type). And many trade studies have been completed without closing the debate on these options. The small LFR studies completed in the U.S. have focused on pool-type designs because it appears to be the more compact and simpler design. The Russian experience was all accumulated using force circulated loop designs which facilitates the thermal hydraulic design and assures coolant mixing throughout the system. However, these designs are not nearly as compact and their SVBR-75/100 commercial design is a pool. The concern with a pool design is that the reactor vessel may become more complicated and the in-vessel flow path may not produce the mixing that would assure uniform coolant chemistry and well understood temperature distributions. Modern CFD codes make it possible complete high confidence detailed thermal hydraulic analysis to support a pool design that should help reduce uncertainties about the T/H design. The choice of pool or loop design does not have a big impact on the issue of an early demonstration being possible. Either configuration could be implemented.

The coolant can be either force circulated using mechanical pumps (EM-pumps are also a possibility but will be very inefficient with Pb or LBE) as in the SBVR-75/200 or naturally circulated as in the current SSTAR design. The natural circulation design is preferred from the standpoint of simplification, reliability and cost, however it does introduce three issues that need to be addressed. First, the added height and coolant inventory needed to achieve full power natural circulation, along with the open lattice large diameter fuel pins required to reduce pressure drop need to be further evaluated. Second, an acceptable approach to startup from a very low flow condition needs to be established. And third, a flow path design that assures coolant mixing and uniform distribution of oxygen in the coolant so that the coolant chemistry I&C can support assuring proper oxygen control throughout the coolant volume.

Natural circulation cooling of the shutdown reactor must be assured in both pool and loop designs. And an air cooled RVACS capable of performing the normal shutdown and inherent shutdown cooling within acceptable temperature limits must be provided. This can be accomplished with the taller vessel used in the naturally cooled system. Based on the SVBR-75/100, that uses a water pool around the reactor vessel, natural circulation air cooling may not be possible with the more compact force circulated system.

These issues can be addressed during the detailed design and do not significantly impact the possibility of completing an early demonstration LFR. The major risk facto concerns the problem of assuring oxygen control throughout the coolant.

3.10 Power conversion system

As mentioned in Section 3.1 the balance of plant should interface with the reactor so that the in-vessel heat exchanger and the system transients can be prototypical of the commercial design. This is most directly accomplished by including a turbine generator system. An air dump heat exchanger can be considered if it can achieve simulation of the operational transients. The choice of system is also dependent on the selected reactor temperature. The SCO_2 system included in current reference would need to be evaluated for its applicability at the desired lower temperature. There appears to be a possibility of using the SCO_2 system with a reactor limited to 550°C peak cladding temperature.

A steam-water power conversion is the most conventional approach and could be either saturated or super-heated steam. The lower primary system temperature will not support a super-critical cycle. The saturated steam system would be the most straight-forward system with little risk. In this case, the in-vessel heat exchanger can operate without departure from nucleate boiling and the steam can be separated in a steam drum. This is the approach used in the SVBR-75/100 and would achieve about a 32% efficiency. Implementing a superheated once-through steam generator would achieve higher efficiency but could lead to lower reliability. The development of the in-vessel heat exchanger and the plant control systems would be more of a challenge with the once-through superheat steam. If these challenges were determined to not impact the cost and schedule, beyond that required for the reactor, the higher efficiency system is preferred. Either steam system could be connected to an air-dump heat exchanger if it would reduce cost and support the transient testing.

A gas cooled system using helium or air is also possible but this option has not been considered for SSTAR. CRIEPI has evaluated both a closed cycle He gas turbine and an open cycle air turbine power plant for 4S. Either of these could be applied to the small lead or LBE cooled reactor but the efficiency in both cases is about 25%. In the case of the helium the additional problem of helium leakage and supply becomes an issue. The simplification provided by the open cycle air system could be attractive and should be considered in the commercial design option prior to making a decision on the early demonstration design. The unattractive low efficiency may be balanced by the high reliability and simplifications in such a system. This approach should be evaluated before making a decision on the technology demonstration.

In general, the selection one of these options does not impact the feasibility of an early demonstration of the LFR. Either steam or gas can be used and the SCO_2 may also be possible. The choice depends primarily on the power conversion system most likely to be used in the commercial design and secondarily on the cost and schedule of the system development. Only the saturated steam system can be implemented without any R&D. The gas turbine system may not require much development but the inlet temperature is lower than conventional turbine design and equipment may not be readily available. The once-through superheated system is likely to require some R&D related to the steam generator and system control. The SCO_2 system, if practical at lower temperature, will clearly require substantial R&D and as a result will introduce greater programmatic risk.

3.11 Instrumentation & Control (I&C)

The instrumentation, with the exception of the coolant chemistry I&C are essentially the same as those developed for sodium cooled fast reactors. The reactor control mechanisms will have some unique issues to address because of the coolant density. Most of the materials commonly used for reactor control are less dense than lead or LBE control materials and therefore the mechanisms for moving and retaining control elements in position will be different than commonly used. There may be design features in the SVBR-75/100 that can be implemented without major development and testing.

There has been I&C experience developed with the DELTA loop, particularly important is the I&C used to control the oxygen concentration in the coolant. There will be a need to expand this effort to develop and demonstrate the adequacy of the oxygen control system for the plant.

Both the neutron control system design and the coolant chemistry control system are critically important development efforts and could be on the critical path for a technology demonstration. It may be necessary to develop two diverse reactor control mechanisms if the inherent safety capabilities are determined to not be sufficient to support a single design. This has not been done for the SVBR-75/100. Both areas deserve early attention.

3.12 In-service Inspection and Maintenance

The in-service inspection requirements for the lead or LBE reactor are very similar to that of a sodium cooled reactor. The sodium cooled reactor experience should support design of most of the features and equipment needed to implement periodic inspection and routine maintenance. Because of the density difference between sodium and lead or LBE it may require some revisions to the sodium cooled reactor experience. It should be possible to complete any engineering development on a schedule to support the near-term LFR demonstration.

3.13 Siting and Safety Review

The early technology demonstration is not envisioned as a project that would be followed shortly by an application for NRC standard design approval or construction of a commercial plant. Therefore the extent of the NRC involvement is open to many considerations. If commercial funding were available it could be constructed on a private site and licensed by the NRC either as a power reactor or a test reactor. In this case the EIS and the NRC review could be conducted using either 10CFR50 or 10CFR52. There is no technical or regulatory rule that would exclude such an approach. If the federal government funds the project it would be expected that the plant would be constructed at INL or another DOE site. In this case, because of the very low power level, it will be a DOE management decision as to what extent the NRC should be involved. The NRC involvement may also be determined by congress during the funding approval process.

There would be greater flexibility to address the unique developmental characteristics of the project using the review process under DOE procedures. It would also remove the need to educate the NRC staff and possibly revise their regulations so that they could be applied to lead or LBE technology. If the project is completed entirely under DOE regulations the completeness of the technology demonstration would be reduced in that an NRC license would not be obtained. The rigor and completeness of the reviews would not be expected to be different in each case but procedural elements, for example, involvement of the ACRS, and the public participation would be different.

Any of these paths could be followed and all would require completing selected R&D in areas identified above. Since the technology demonstration is the first-of-a-kind and will require some development to be completed, it is preferable to proceed along the lines of the FFTF project where there was NRC participation, including the ACRS, but no NRC license was requested.

4.0 Possible Paths to an Early LFR

This section discusses two possible program levels and the options that are available to reduce the cost and risk. These two programmatic options have been developed based on the evaluations in Section 3.0.

4.1 Minimum Scope

The minimum scope project should demonstrate at a minimum the ability to reliably accommodate the unique characteristics of LBE coolant and provide irradiation facilities to support development of advanced fuels and materials for future commercial designs. A reactor of about 60MWt would be constructed. To the extent possible the project should develop information and experience that would support the possibility of U.S. industry becoming a supplier of a commercial small LFR design. This objective eliminates the possibility of purchasing the SVBR-75/100 and installing it in an existing DOE facility, which may be the lowest cost approach for gaining operational experience with an LBE system. Although much would be learned by the approach, there would be many unknowns remaining and it would not likely be possible for U.S. industry to become a supplier based on this experience unless it partnered with the Russians, an unlikely possibility.

The facility should also be capable of demonstrating the operational and safety performance required in a commercial plant design and thus the reactor core and primary coolant path should be as prototypical as possible. The reactor should be designed and approved for operation on a DOE site for a period of time sufficient to obtain the fuels and materials data needed to support a commercial design. It is judged that 15 years of operation should be long enough to obtain the necessary data. Such an operation would not demonstrate the target 30 year plant life nor would all of the features that are necessary for the commercial design be included. It should provide sufficient neutronic, reactor coolant T/H and structural performance data necessary to calibrate the design tools needed to produce the commercial design with 30 year life, or confirm an alternative acceptable lifetime.

In addition to the neutronic, reactor T/H and structural data the project will provide the following:

1. Provided irradiation capabilities for advanced fuels and material development
2. Selected manufacturing, construction experience, shipping and installation procedures for projected commercial sites to the extent inherent in this limited scope, i.e. no extra effort would be extended beyond what is needed to obtain the test facility
3. Methods for refueling based on single assembly replacement. This will not demonstrate the cassette core, or whole reactor replacement options but will provide data useful to design of these options.
4. Provide facilities and equipment for handling and storing spent fuel and material irradiation fixtures. These are not required in the commercial design but will be needed in this facility to accommodate the experimental programs. It will need to be determined whether or not in-vessel storage is required and what type of ex-vessel storage will be provided.
5. Demonstrate the ability for autonomous reactor operation, including automated startup and shutdown and load following over a power range yet to be specified. Exactly how this is done will depend on the final selection of the balance of plant. For use of saturated steam cycle with a turbine generator then testing would be straight forward. If a heat dump is used because it reduces cost then it may be an incomplete demonstration.
6. Demonstrate the capability to accommodate selected unprotected transients. This may be accomplished at the end of the program because the temperatures reached may impact the core or structural life.
7. Demonstrate reliability of key reactor components for use in commercial design. This means the core components and reactor structure. Prototypical equipment for the purposes of reliability demonstration that would increase cost will not be included.
8. Demonstrate reactor instrumentation and controls suitable for the commercial design. Balance of plant or facility equipment will be selected to achieve effective performance from the facility and not necessarily be protoyical.

9. Demonstrate equipment and operations for reactor coolant chemistry control including systems to accommodate Po-210.

4.2 Extended Scope

The extended scope of effort should develop information and experience that would support the possibility of U.S. industry becoming a supplier of a commercial small LFR design. The project would include the effort for design of a commercial plant operating on super-heated steam. It would include once-through steam generators that may require development testing to support the design. The power level would be selected based on interest of a near-term application but would not exceed 125MWt. An extended effort would be included to make most of the plant prototypical of a commercial design. The plan would remain to operate the plant for 15 years.

In addition to the items identified in the minimum scope:

1. The manufacturing, shipping and site installation would be completed as anticipated for a commercial site
2. Balance of plant will be prototypical of steam powered commercial design
3. The core and core support would be designed to demonstrate the commercial plan for no on-site storage of fuel outside the reactor. Fuel and test assembly storage facilities would be provided but the commercial approach for fuel management would also be demonstrated. Exactly how this is implemented will depend on design. The design may include a cassette core for whole core replacement and equipment to remove and place it in a fuel storage facility or shipping container. However, there are several alternatives, including single assembly transfers to multiple assembly shipping casks. If the cassette design is used it would include locations for irradiation testing of advanced fuels and materials.

5.0 Preferred Path

5.1 Reactor Design Options

Two options have been identified based on power level and power conversion system, a small 60MWt plant using a saturated steam cycle and a larger 125MWt using a super-heat steam cycle. Based on the U.S. LFR work performed to date, there is a strong preference for either option to use natural circulation because the simplification and increased reliability this approach would provide. The natural circulation cooling also requires a reactor vessel size that is more compatible with an air-cooled RVACS and avoids the water cooling jacket used in the SVBR-75/100.

From the standpoint of performance, the larger plant would be preferred but there are likely to be significant cost savings with the smaller plant and it can probably meet the minimum objectives of the technology demonstration. It would demonstrate the viability of the LBE coolant and corrosion control in a reactor system. It would also provide some capability for fuel and materials development testing. A detailed trade study on the core size and number of test locations is needed to establish the minimum acceptable size. Given the premium placed on placed on cost and schedule the smallest system that meets the demonstration objectives is preferred and this is likely to be achieved at about 20MWe.

It is judged that there is not likely to be much cost advantage from elimination of the power conversion in the balance of plant given that the generated electricity will have value, and it is not clear that a heat dump system that could simulate the transient testing would necessarily be cheaper.

The super-heated steam cycle or SCO_2 system would require some engineering development beyond the saturated cycle but either one may provide both configuration (avoiding steam drums) and performance advantages that could balance the cost of the development. Detail evaluation of these options should be completed.

5.2 Program Scope

The scope outlined in Section 4.1 should be implemented and the safety review conducted using DOE Regulations and Directives supported by NRC oversight, including participation of the ACRS, like was done with FFTF. The developmental status of a first-of-a-kind LBE cooled reactor would place excessive burden on the NRC if the facility were to be licensed under 10CFR50. The flexibility provided in the DOE Regulations and Directives would permit the project to proceed to develop the necessary new safety criteria as the project proceeds. It would be very time consuming, but not impossible, to conduct the safety development in parallel with NRC licensing. It appears unnecessary to taken the burden of doing development in parallel with licensing given that the first commercial plant following the demonstration project is far enough in the future to permit a systematic approach to the safety development and the NRC preparation. NRC oversight would benefit both the project and the NRC by preparing the NRC staff for future LBE cooled reactor licensing projects and providing the feedback to the demonstration project that would assure that the safety development addresses licensing issues that might otherwise be omitted.

5.3 ROM Cost

A rough order of magnitude ROM cost for the minimum scope outlined in Section 4.1 has been developed from several sources of information. Given the status of LFR development it should be recognized that this estimate is subject to much uncertainty.

The Russians have estimated the cost of constructing and completing the R&D for a prototype SBVR-75/100, if installed in an existing nuclear facility in Russia, to be $100 million. The specifics about this estimate are not available, but it is known that considerable engineering of this design has been completed. Therefore if one is to start from a conceptual design there is a cost delta simply to bring the design to the current level of the SBVR-75/100. In addition, the labor rates in Russia compared to those in the U.S are between 3 and 5 times less. Putting these factors together one might add $100-200 million for the design and $300-500 million for the construction. Based on these assumptions, the cost of a 100MWe prototype LFR in the U.S. might be between $400-700 million. Scaling construction down to a 20MWe (60MWt) that could be used in a technology demonstrator, using a 0.4 scaling factor, the estimated cost would be reduced to $260-460 million.

There has been discussion of constructing a small 4S, 10MWe, sodium cooled reactor in Galena Alaska []. The cost of the plant has been reported to be $20 million. Assuming this estimate is for the equipment cost without fuel and that the equipment costs are 1/3 of the total cost one would estimate a total cost of $60 million plus the cost of the fuel. Scaling up the cost for a $20MWe plant would be about $80 million. However, this is also for a sodium cooled plant that may require less engineering development and has had considerable engineering effort applied to its design. Still, based on this information it seems reasonable to think the minimum scope LFR could be constructed for approximately $100 million, considerable less than the estimate based on extrapolating SVBR-75/100.

LLNL has completed a cost study for the small LFR using the detailed cost estimate for the ALMR [] as a source of information. The result of this effort indicates that the capital cost for the 50 MWe LFR is between $3500-4500 per kwe for the Nth of-a-kind plant, depending on the cost of the fuel. The total cost is $175-225 million and if scaled down to 20 MWe the total cost is approximately $120-160 million. This is based on mature Nth of-a-kind experience and therefore would be increased for the LFR demonstrator. This cost difference is likely to be less than 50% based on learning curve and factory assembly. Therefore based on this study one might expect a cost of $180-$240 million.

Table 5-1 summarizes these capital cost estimates and adds the cost of operating for 15 years at a cost of $30 million per year. (This corresponds to an average of approximately 85 people per day at $40/hr.) The range of costs for the 20 MWe minimum scope project based on various estimates of the capital cost are seen to be between $550-910 million. If the estimates for the total costs are averaged it the result is $698 million which is the best estimate.

Table 5-1 ROM Cost Estimate for Minimum Scope LFR, Millions of Dollars

Source Data	Capital Cost	15 year Operating Cost	Total Project Cost
SBVR-75/10	260-460	450	710-910
Galena 4S	100	450	550
LLNL Cost Study	180-240	450	630-690

Cost of the extend project would be expected to double, at a minimum and would also introduce greater programmatic risk.

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